NON-PUBLIC?: N
ACCESSION #: 8903290172
LICENSEE EVENT REPORT (LER)

FACILITY NAME: NORTH ANNA POWER STATION, UNIT 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000338

TITLE: REACTOR TRIP DUE TO MAIN FEEDWATER REG. VALVE CLOSURE AND SUBSEQUENT
S/G TUBE LEAK
EVENT DATE: 02/25/89 LER #: 89-005-00 REPORT DATE: 03/23/89

OPERATING MODE: 1 POWER LEVEL: 76

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(i), 50.73(a)(2)(iv), 50.73(a)(2)(v)

LICENSEE CONTACT FOR THIS LER:
NAME: G. E. Kane, Station Manager TELEPHONE: 703 894-5151

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: FCV MANUFACTURER: C635
X AB SG W120
X SB FCV F130
X IL RA W120
X IG DET W120
X IG DET W120
X WI ISV F130
X SJ FCV C635
X IG RLY
X IG DET W120
REPORTABLE TO NPRDS: Y
Y
Y
N
Y
Y
Y
N
Y
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1407 hours on February 25, 1989, Unit 1 automatically tripped from 76 percent power (Mode 1). The initiating signal for the reactor trip was "C" Steam Generator steam flow greater than feedwater flow mismatch coincident with a low Steam Generator level. The
team flow greater than feedwater flow
mismatch was caused by the closure of the "C" Main Feedwater Regulating Valve on the loss of control air. This event is reportable pursuant to 10CFR50.73(a)(2)(iv).

Following the reactor trip, indications of primary to secondary leakage were detected. "C" Steam Generator was identified as the source of the leakage. The Emergency Plan was entered and an ALERT was declared. The plant was subsequently cooled down and depressurized to Mode 5 and the ALERT was terminated. During the cooldown, a problem was encountered placing the Residual Heat Removal System in service due to a faulty auto-closure relay on the RHR Suction Isolation Valve (1-RH-MOV-1701). This is reportable pursuant to 10CFR50.73(a)(2)(v). Following the event, calculations were made that indicated that primary the secondary leakage was 74 GPM. This is reportable pursuant to 10CFR50.73(a)(2)(i)(A).

This event posed no significant safety implications because safety equipment functioned as designed. Radiological releases were well below Technical Specification limits. The health and safety of the general public were not affected during this event.

END OF ABSTRACT

Component Failure Description included on LER form.

1.0 Description of Event

At 1407 hours on February 25, 1989, Unit 1 automatically tripped from 76 percent power (Mode 1). The initiating signal for the reactor trip was "C" Steam Generator (S/G) (EIIS System Identifier AB, Component Identifier SG), steam flow greater than feedwater flow mismatch coincident with a low S/G level. The steam flow greater than feedwater flow mismatch was caused by the closure of the 'C' Main Feedwater Regulating Valve (EIIS System Identifier SJ, Component Identifier C635, Model Number D100). The "C" Main Feedwater Regulating Valve closed due to the fatigue failure of the instrument air supply line around the fitting on the valve positioner.

This event is reportable pursuant to 10CFR50.73(a)(2)(i). A four hour report was made in accordance with 10CFR50.72(b)(2)(ii).

Following the reactor trip, Reactor Coolant System (RCS) pressure and temperature decreased to approximately 1950 psig and 545 Degrees F, respectively. RCS temperature was then stabilized at 547 Degrees F by automatic operation of the steam dumps. The Auxiliary Feedwater Pumps (EIIS System Identifier BA, Component Identifier P) started automatically due to Low-Low Steam Generator level. It was then noticed that RCS pressure was not increasing as fast as expected and more charging flow than expected was needed to restore and maintain pressurizer level. Pressurizer level was raised and maintained at 20% and letdown was restored three minutes after it isolated on low pressurizer level. 1-EP-0, Reactor Trip or Safety Injection, was entered. In accordance with this procedure, plant parameters were evaluated and no conditions requiring a safety injection existed. 1-ES-0.1, Reactor Trip Response, was then entered as required by 1-EP-0.

At 1426 hours, the air ejector Radiation Monitor 1-RM-RMS-121 Hi and Hi-Hi alarms were received. 1-AP-5.1, Unit 1 Radiation Monitoring System, was entered, and the air ejector discharge was verified to swap to containment on the Hi-Hi radiation alarm. Health Physics was directed to sample the air ejector discharge, and the sample results indicated a significant activity increase. Chemistry personnel were then requested to obtain samples from each Steam Generator blowdown line to identify increases in activity. The blowdown system had automatically isolated on Low-Low Steam Generator level following the reactor trip. Once Steam Generator level was restored, realignment of the blowdown system was made to obtain the sample. Subsequent to the realignment, "C" Steam Generator Blowdown Radiation Monitor increased to the Hi-Hi alarm setpoint. The "C" Main Steam Line Radiation Monitor showed a slight increase in activity.
As a result of the necessity to increase charging flow to maximum with one charging pump in order to maintain RCS inventory and the various (although erratic) alarms on the radiation monitors, Abnormal Procedure 1-AP-24.1, Large Steam Generator Tube Leak Requiring Immediate and Rapid Unit Shutdown was entered.

After "C" Steam Generator was identified as the source of primary to secondary leakage, emergency boration was started and the "C" Steam Generator steam supply valve to the steam driven auxiliary feedwater pump was manually isolated in accordance with 1-AP-24.1.

Based on charging and letdown flow rates, primary system leakage was estimated to be 60-70 GPM. The Station Emergency Manager declared an

ALERT at 1525 hours based on a RCS leak of greater than 50 gpm. This was a conservative action because the Emergency Plan Implementing Procedures (EPIPs) require an ALERT to be declared when RCS leakage is greater than 50 gpm and more than one charging pump is needed to maintain pressurizer level. (Only one charging pump was needed to maintain pressurizer level.) Notification to the state and local governments was completed by 1539 hours. Notification to the NRC was made at 1556 hours as required by 10CFR50.72(a)(1)(i). The Emergency Response organization was established in accordance with the EPIPs. Primary to secondary leakage greater than Technical Specification 3.4.6.3 limits is reportable pursuant to 10CFR50.73(a)(2)(i)(A).

A RCS cooldown commenced at 1526 hours in accordance with 1-AP-24.1. At 1600 hours RCS pressure was reduced to below "C" Steam Generator pressure and the level in "C" Steam Generator started to decrease. Maximum level reached was 72% narrow range level. 1-ES-3.1, Post Steam Generator Tube Rupture Cooldown Using Backfill, was then entered to cooldown and depressurize the RCS to Mode 5.

The plant entered Mode 4 at 1833 hours. While placing the Residual Heat Removal (RHR) System in service as per 1-OP-14.1, Residual Heat Removal System, RHR Suction Isolation Valve (1-RH-MOV-1701) (EIIS System Identifier BP, Component Identifier ISU) failed to remain open after reaching its full open position. Once full open indication was received, the valve immediately stroked closed. The RHR System could not be placed in service due to this malfunction. A four hour report was made in accordance with 10CFR50.72(b)(2)(iii)(B). This event is reportable pursuant to 10CFR50.73(a)(2)(v). Troubleshooting revealed that the high RCS pressure auto-closure relay (EIIS System Identifier JG, Component Identifier RLY, Vendor Identifier W120) caused a close signal to be generated when the valve reached the full open position. The auto-closure relay was defeated long enough for power to be removed from the valve operator, with the valve in the full open position. The redundant RHR suction valve, 1-RH-MOV-1700, stroked full open with no problems. The RHR System was placed in service at 2143 hours.

RCS cooldown continued, and the plant entered Mode 5 at 2212 hours. At 2220 hours, the ALERT was terminated. The NRC was notified of event termination at 2222 hours and state and local governments were notified at 2227 hours. Following the event, primary to secondary leakage was calculated to have been 74 GPM.

Plant equipment responded as expected with the following exceptions:

"C" Main Feedwater Regulating Valve, 1-FW-FCV-1498, failed closed due to a broken air line.

"B" Main Feedwater Regulating Valve, 1-FW-FCV-1488, (EIIS System Identifier SJ, Component Identifier FCV, Vendor Identifier C635) indicated mid-position when closed.

"C" Steam Generator, 1-RC-E-1C, (EIIS System Identifier AB, Component Identifier SG, Vendor Identifier W120) developed a tube leak due to the failure of a mechanical tube plug.

Reheat steam flow control valve to the "D" Moisture Separator Reheater, 1-MS-FCV-104D, (EIIS System Identifier SB, Component Identifier FCV, Vendor Indentifier F130) did not close on demand from the control room.

Condenser Air Ejector Radiation Monitor, 1-RM-SV-121, (EIIS System Identifier IL, Component Identifier RA) exhibited erratic behavior and water was found in the vent discharge.

The Intermediate Range Nuclear Instrumentation (NI) (EIIS System Identifier IG, Component Identifier DET, Vendor Identifier W120) was under compensated, and the Source Range Nuclear Instrumentation had to be manually energized, in accordance with procedure.

Source Range Neutron Flux Monitor 1-NI-N32 (EIIS System Identifier IG, Component Identifier IG, Component Identifier DET, Vendor Identifier W120) detector failed.

"C" Steam Generator Blowdown Line inside containment Isolation Valve 1-BD-TV-100F (EIIS System Identifier ISU), did not close.

The plant computer did not document a flux rate trip from Power Range Neutron Flux Monitor 1-NI-N41 (EIIS System Identifier IG, Component Identifier DET, Vendor Identifier W120).

Residual Heat Removal System Inlet Valve 1-RH-MOV-1701, failed to stay open.

2.0 Significant Safety Consequences and Implications

This event posed no significant safety implications because safety equipment functioned as designed.

Environmental air samples and TLDS collected after the event did not indicate any measurable increase in radioactivity or radiation levels at the monitoring locations.

Attachment 1 provides the analysis of the radiological data from the very low releases to the environment during the event. As can be seen, radiological releases were well below Technical Specification limits. The pathways for the releases were the discharge of the Condenser Air Ejector and the Steam Driven Auxiliary Feedwater Pump exhaust. The health and safety of the general public were not affected during the event.

3.0 Cause of the Event

A root cause analysis has determined that the "C" Main Feedwater Regulating Valve closed due to fatigue failure of the instrument air supply line around the fitting on the valve positioner. The fatigue was induced by vibration of the main feedwater regulating valve during normal operation.

The "C" Steam Generator tube leak was caused by the failure of the hot leg tube plug located at Row 3, Column 60. This tube was plugged in November 1985 as part of the normal ISI Program. Examinations will be conducted on the plug to determine the cause of the failure.

The cause for the RHR Suction Isolation Valve, 1-RH-MOV-1701, failing to remain open during cooldown was due to the failure of the high pressure auto-closure relay. This failure generated a close signal when the valve reached the full op
n position.

4.0 Immediate Corrective Action

As an immediate corrective action, Unit 1 was placed in Mode 5, in accordance with station procedures.

5.0 Additional Corrective Actions

An action plan has been developed to investigate equipment malfunctions that occurred during the reactor trip and subsequent cooldown. Corrective actions are being implemented.

The failed mechanical plug in "C" Steam Generator, along with three additional mechanical plugs, have been sent to Westinghouse to perform a failure analysis.

6.0 Actions to Prevent Recurrence

Based on the results of a root cause evaluation, air lines associated with the Unit 1 and 2 Main Feedwater Regulating Valves will be modified, as necessary, to reduce the possibility of fatigue failures.

Investigation of the Steam Generator tube plug failure is ongoing. Actions to prevent recurrence will be based on the results of the investigation. A separate and more detailed report will be provided on the Steam Generator Tube Leak event. Preliminary indications are that the tube plug failed due to primary water stress corrosion cracking (PWSCC).

7.0 Similar Events

Previous reactor trips due to steam flow greater than feedwater flow mismatch coincident with a low steam generator level occurred on Unit 1 August 6, 1988, (LER-NI-88-020-00) and May 20, 1986 (LER-NI-86-008-00) and on Unit 2 on March 13, 1984 (LER-N2-84-001-00), June 25, 1984, (LER-N2-84-005-00) and on June 29, 1986 (LER-N2-8#8-009).

On July 15, 1987 (LER-NI-87-017-01), a tube rupture occurred in the Unit 1 "C" Steam Generator due to fatigue failure.

8.0 Additional Information

It was decided to enter a planned refueling outage before returning Unit 1 to service. Operating experiences gained during this event have been made available to the industry by means of INPO's Nuclear Network System.

North Anna Unit 2 was in Mode 5 for a scheduled refueling outage, and was not affected during this event.

ATTACHMENT 1

ANALYSIS OF RADIOLOGICAL DATA

Releases of noble gases, radioiodines, particulate and tritium were evaluated for applicable airborne pathways. The following activities were calculated to have been released:

Noble Gases: 5.28E+0 Ci

Radioiodines: 9.44E-8 Ci
Particulates: 4.54E-8 Ci
Tritium: 1.00E-2 Ci

Dose rates, calculated based on estimated maximum release rates, were well within Technical Specification allowable dose rates at and beyond the site boundary:

Tech Spec Dose Rate Limit Event Dose Rate % Tech Spec

Total Body, 500 mrem/yr 1.2E+1 mrem/yr 2.3E+0
Skin of Whole Body, 3000 mrem/yr 2.4E+1 mrem/yr 8.2E-1
Critical Organ, 1500 mrem/yr 3.0E-2 mrem/yr 2.0E-3

Doses, calculated based on estimated total activity released, were well within Technical Specification limits for doses at and beyond the site boundary:

Tech Spec Dose Rate Limit Event Dose Rate % Tech Spec

Noble Gas Gamma Air Dose, 10 mrad 2.0E-3 mrad 2.0E-2
Noble Gas Beta Air Dose, 20 mrad 2.7E-3 mrad 1.4E-2
Critical Organ Dose, 15 mrad 1.1E-6 mrad 7.3E-6

ATTACHMENT 1 TO 8903290172 PAGE 1 OF 1

Vepco 10 CFR 50.73

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION
P. O. BOX 402
MINERAL, VIRGINIA 23117

March 23, 1989

U. S. Nuclear Regulatory Commission Serial No. N-89-010
Attention: Document Control Desk NO/JJM: nih
Washington, D.C. 20555 Docket No. 50-338

License No. NPF-4

Dear Sirs:

The Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Unit 1.

Report No. LER 89-005-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to Safety Evaluation and Control for their review.

Very truly yours,

G. E. Kane
Station Manager

Enclosure

cc: U. S. Nuclear Regulatory Commission
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. J. L. Caldwell
NRC Senior Resident Inspector
North Anna Power Station

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